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444 South 16th Street Mall
Omaha NE 68102-2247

May 5, 2003
LIC-03-0059

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Reference: Docket No. 50-285

Subject: Licensee Event Report 2003-001 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 2003-001, Revision 0, dated May 5, 2003. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B).

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/EPM/epm

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
INPO Records Center
Winston and Strawn

IE22

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Lack of Guidance Results in Noncompliance With Technical Specification Surveillance Requirement

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	06	2003	2003	- 001 -	0	05	05	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Charles N. Bloyd	402-533-6921

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

During the week of March 3, 2003 an evaluation to determine the adequacy of the Fort Calhoun Station (FCS) boric acid program was conducted. As part of the evaluation, one of the evaluation team members requested information on the results of the VT-2 inspections on the lower portion of the reactor vessel. A review of the inspection results indicated that the VT-2 examination had not been accomplished which is not in compliance with Section XI of the ASME Boiler and Pressure Vessel Code as required by Technical Specification section 3.3.1.a.

The most probable cause of this event is lack of procedural guidance, caused by poor human factors in the FCS procedure that is used to inspect the rest of the reactor coolant system. There appears to have been a mind set among individuals that the room housing the reactor vessel was then, and had always been, inaccessible because of radiological dose considerations. Further, there appears to be a mind set that "inaccessible" because of radiological dose considerations is equivalent to "inaccessible" as defined in the code.

FCS will perform the examination as required by the code if the relief request submitted for processing prior to the 2003 refueling outage is delayed or denied.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

BACKGROUND

The Fort Calhoun Station (FCS) reactor coolant system (RCS) consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two reactor coolant pumps, connecting piping and instrumentation. A pressurizer is connected to one of the reactor vessel outlet (hot leg) pipes by a surge line. Pressurizer relief and safety valves are provided which discharge to a quench tank to condense and cool valve discharges. All components of the RCS are located within the containment building.

The RCS is designed to remove heat from the reactor core and internals and transfer it to the secondary (steam generating) system by the controlled circulation of pressurized, borated water which serves both as a coolant and a neutron moderator. The RCS serves as a barrier to the release of radioactive material to the containment building and is equipped with controls and safety features that ensure safe conditions within the system. The design pressure is 2500 psia, design temperature 650F (pressurizer 700F),and design life 40 years.

The FCS reactor vessel has an integral lower head (i.e., there are no mechanical penetrations in the bottom of the FCS reactor pressure vessel). The FCS reactor vessel does not have any penetrations below the reactor coolant inlet and outlet nozzles. As required by the code an inspection of the reactor vessel welds was conducted in 1992 with no adverse results noted (discussed in greater detain below).

EVENT DESCRIPTION

During the week of March 3, 2003, an evaluation to determine the adequacy of the FCS boric acid program was conducted. As part of the evaluation team members requested information on the results of the VT-2 inspections on the lower portion of the reactor vessel. A review of the inspection results indicated that the VT-2 examination beneath the reactor vessel had not been accomplished which was not in compliance with Section XI of the ASME Boiler and Pressure Vessel Code as required by Technical Specification section 3.3.1.a. FCS Technical Specifications section 3.3.1.a. requires: "In-service inspection of ASME Code Class 1, Class 2, and Class 3 components, including applicable supports, and in-service testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a (g)(6)(i)."

The VT-2 inspection for the RCS is conducted using Surveillance Test, OP-ST-RC-3007, "Reactor Coolant System Integrity Test Following Opening, Repair or Modification". (Revision 12 was the revision used during the 2002 Refueling Outage.) This test seeks to comply with Section XI in two distinct steps. Step 7.7.1 states, "QC perform VT-2 visual examination on each RCS component listed in Attachment 2. Attachment 2 lists components that are typically opened during the course of a cold shutdown, such as, manways and bolted connections. Step 7.7.2 states, "QC perform VT-2 visual examination on all RCS Class I piping including, but not limited to, those associated with the following list:

- Reactor Coolant Pumps (all)
  - Seals & Seal Piping
  - Case to cover flange
- Steam Generators (both)
  - Primary Manways
- Pressurizer (including heaters)
  - Manway
  - Instrumentation Piping
  - Heaters

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- Reactor
  - Head to Vessel Flange
  - ICI Grayloc Flanges
  - CEDM Tool Access Flanges
  - CEDM Seal Housing Flanges"

OP-ST-RC-3007 step 7.7.3 states, "Perform VT-2 visual inspection on all Class I piping associated with the items listed in Attachment 3". Attachment 3 lists numerous specific valves associated with the RCS or associated systems. Nowhere in OP-ST-RC-3007 is there a specific requirement to perform a VT-2 visual inspection on the reactor vessel after a four hour "soaking" period. There are specific instructions to perform VT-2 inspections of unique locations of the RCS.

Earlier versions of the OP-ST-RC-3007 were reviewed. Revision 0 has an equivalent step to step 7.7.2 of revision 12 that states, "Inspect all RCS Class I components and piping including, but not limited to, those associated with the following:

- Reactor Coolant Pumps (all)
- Steam Generators (both)
- Pressurizer (including heaters)
- Reactor"

In May, 1990, there was an effort to ensure that Inconel 600 penetrations were emphasized during the VT-2 inspections. Therefore, under each piece of major equipment listed in the step reflecting the four hour hold, subsections were added indicating specific portions of the major equipment in an attempt to improve OP-ST-RC-3007. Revision 2 was issued in an attempt to further clarify and standardize OP-ST-RC-3007. The step was changed, and the word "component" was apparently inadvertently left out of the step. The step only required the inspection of piping.

On March 7, 2003, a review of the reportability of this event was completed. This is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B).

SAFETY SIGNIFICANCE

This issue has a negligible impact on the health and safety of the public as discussed below:

There is little possibility of an unacceptable flaw or flaw propagation in the region of the reactor vessel not examined by the surveillance test. A visual examination would only have detected a through-wall flaw, and would have provided no additional assurance for less than through-wall flaws of any size. This conclusion is supported by the following:

Possibility of Undiscovered Flaw
The presence of an undiscovered flaw resulting in leakage from the reactor vessel is not probable. The portion of the reactor shell in question does not contain any materials other than the reactor shell material (i.e., there are no mechanical penetrations in the bottom of the FCS reactor pressure vessel). That is, there are no materials present that are susceptible to rapid crack initiation and propagation mechanisms under the reactor operating conditions, such as the Ni-Cr-Fe alloy 600 family of materials. Consequently, there is considered a low likelihood of an active cracking mechanism that would produce flaw initiation and growth.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

This portion of the reactor shell is remote from nozzle penetrations and other discontinuities that would tend to amplify any cyclic loading experienced by the shell (stress intensification), and so it is unlikely that a flaw would initiate or grow by a fatigue mechanism.

Previous examinations, including pre-service examinations, have not identified any significant fabrication-related defects in this area that might subsequently have grown to an unacceptable size by other mechanisms, such as described above. Because any such flaw would have experienced full reactor operating pressure and temperature, both of which would have tended to open a hypothetical through-wall flaw, the absence of any observed leakage products (e.g., boric acid crystals) is evidence that no such flaw actually exists.

Possibility of Flaw Propagation
It is not expected that a hypothetical flaw would propagate to an unacceptable depth. In general, flaw propagation could occur in this type of material by a fatigue mechanism, by an environmentally driven mechanism such as stress corrosion cracking, or by brittle or ductile flaw growth due to shock loading.

In the present case, none of these mechanisms is expected to be active in the Fort Calhoun lower shell region, for the reasons summarized above. The region in question is only subject to operating temperatures and pressures typically. Although the stresses associated with these conditions are of significant magnitude, the plant experiences only a small number of such cycles. Because there are few or no structural discontinuities in this region to intensify the operating stresses, propagation by fatigue cracking is not considered likely.

Stress corrosion cracking requires the presence of a susceptible material for propagation. Such materials are not present in the shell region of interest, and so propagation of a hypothetical flaw by that mechanism could not happen.

The reactor shell material exhibits high toughness at room and operating temperatures. Some types of events (e.g., a pressurized thermal shock event) could conceivably lower the toughness during the event due to the rapid cooling of the shell skin material (the surface immediately in contact with the cold pts event fluid). Although the effect might be to promote brittle crack propagation of a hypothetical flaw initially, the effect is very shallow in extent (that is, the effect falls off very quickly with increasing depth from the wetted surface). A hypothetical flaw would very quickly run out of the temporary low toughness material into the bulk shell material, which has much higher toughness. When this happened, the hypothetical flaw would arrest as the applied stress intensity factor fell below the arrest toughness, Kia, of the material, as described in ASME section XI, appendix A. Such a flaw would then become static, due to a lack of crack driving mechanisms.

Material Toughness
The fluence to the welds and plate material is less than 1.0E17 n/cm2, which is below the NUREG 1801 threshold of evaluation. The single arc girth weld (10-410) is comprised of weld wire heat 13253 which has a chemistry factor of 190.4F. This is a non-limiting chemistry factor versus the beltline welds which have a maximum value of 215.5F. For the longitudinal seam welds (1-411a/f) manual arc welding was performed using stick electrodes and the chemistry factor of these welds is 95F, thus making these welds non-limiting as well.

Critical Crack Size

Preliminary calculations indicate the critical size of a through-wall crack, at normal operating pressure, would be approximately 10" in length. Due to the material properties discussed above, this critical crack size would not be significantly affected by a pressurized thermal shock event. It is reasonable to expect that a through-wall crack of this size would be detected by the normal leak detection processes.

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1992 Reactor Vessel Inservice Inspection
Per ASME section XI requirements, the reactor vessel weld inspection was performed during the 1992 refueling outage. This inspection consisted of 176 automated ultrasonic exams and an interior visual exam of the reactor vessel welds and heat affected zones. The inspection found no rejectable indications in any of the reactor pressure vessel welds or heat affected zones. Twenty minor indications were found that were initially categorized as code recordable.

- twelve (12) of these indications were small laminar indications that did not obscure the backwall UT signal and could be immediately dispositioned as code acceptable.
- two (2) were determined to have code acceptable signal level when re-exam was performed.
- three (3) were found to have been caused by transducer lift-off due to a local surface irregularity.
- three (3) were sized and found to be well within the code allowable acceptance criteria for weld indications.

Based on these results the vessel was found to be sound and satisfactory to be returned to service.

RCS Leak Rate Trending
The FCS staff has assembled 9 cycles of unknown leakage data that demonstrate RCS operating threshold for boundary integrity. This trending of unknown leakage based on a threshold definition and category zones has been verified by previous cycles and validated by known leakage issues. These two categories are grouped into either connection (range from 0.075 to 0.2 gpm) or boundary integrity leakage (greater than 0.2 gpm) in facilitating an assessment of system performance. In monitoring of cycles 19, 20 and 21 (current) it is evident the boundary leakage threshold has not been challenged in the previous and/or current cycle. During the current cycle, the RCS leak rate has been below the threshold baseline.

Therefore, this issue has a negligible impact on the health and safety of the public.

CONCLUSION

The most probable cause of this event is lack of procedural guidance, caused by poor human factors in procedure OP-ST-RC-3007. There appears to have been a mind set among many individuals that the room housing the reactor vessel was inaccessible. When questioned why they thought the room was inaccessible, most individuals stated that it was because of radiological dose reasons. Further, there appears to be a mind set that "inaccessible" because of radiological dose considerations is, in some way, equivalent to "inaccessible" as defined in the code. Some individuals assumed there was an engineering argument that documented the inaccessibility of the room as related to the code. There was an additional argument/mind set from some individuals that, since the word "component" was not included in the test, the reactor vessel inspection was not required. Therefore, it appears that a contributing cause for this event was a number of inappropriate mind sets.

CORRECTIVE ACTIONS

FCS will perform the examination as required by the code if the relief request submitted for processing prior to the 2003 refueling outage is delayed or denied. Other corrective actions related to this issue are being completed in accordance with the FCS corrective action program.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event did not result in a safety system functional failure in accordance with NEI 99-02.

PREVIOUS SIMILAR EVENTS

FCS has not had any previous similar events.